Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

LIC-09-0005
January 23, 2009

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

References:
1. Docket No. 50-285
2. Letter to the NRC (Document Control Desk) from OPPD (T.R. Nellenbach) dated May 13, 2008 (LIC-08-0046)
3. Letter to the NRC (Document Control Desk) from OPPD (R.P. Clemens) dated September 22, 2008 (LIC-08-0097)

Subject: Licensee Event Report 2008-001 Revision 2 for the Fort Calhoun Station

Please find attached Licensee Event Report 2008-001, Revision 2, dated, January 23, 2009. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A). If you should have any questions, please contact me.

Sincerely,

Jeffrey A. Reinhart
Site Vice President
Fort Calhoun Station

JAR/epm

Attachment

c: E. E. Collins, NRC Regional Administrator, Region IV
Alan Wang, NRC Project Manager
J. D. Hanna, NRC Senior Resident Inspector
INPO Records Center

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORYCOMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

Estimated burden per response to complywith this mandatory collection request: 80 hrs. Reported lessons learned are incoporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy S ervice B ranch (T-5 F52), U. S. Nuclear Regulat ory Commission, Washington, DC 20555-0001, or by int emet e-mail t o infocollects@nrc.gov, and to the De sk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Offi ce of M anagement and Budget, Washington, DC 20503. If a means used t o impose an inf ormation collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required t o respond t o, t he information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Fort Calhoun Station	05000285	1 OF 3

4. TITLE: Reactor Trip Due to Turbine Control System Failure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	15	2008	2008	- 001 -	02	01	23	2009		05000
									FACILITY NAME	05000

9. OPERATING MODE: 01

10. POWER LEVEL: 85

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (*Check all that apply*)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	X 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (*Include Area Code*)
Erick Matzke	402-533-6855

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JJ	SC	GE	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (*If yes, complete 15. EXPECTED SUBMISSION DATE*) ☒ NO				

ABSTRACT (*Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines*)

The plant was operating at a reduced power level (nominal 85 percent) due to turbine control system problems. The station was preparing a troubleshooting plan when on March 15, 2008, at 0833 CDT, a reactor trip occurred. The operators entered Emergency Operating Procedure (EOP) 00 "Standard Post Trip Actions." The main steam and feedwater system operated normally. All control rods inserted fully.

The apparent cause of the reactor trip was the failure of a circuit board in the EHC control system resulting in the closure of turbine control valves CV-1 and CV-3. Closure of these valves results in a pre-emptive trip of the reactor due to the loss of load. The root cause of the circuit board failure was an improperly adjusted potentiometer, which resulted in a long term overcurrent condition to one of the transformers on the affected circuit board. The overcurrent condition resulted in the failure of the affected transformer.

The failed circuit board was replaced. The output of the affected potentiometer was verified to be correctly adjusted. Post maintenance testing was performed on the EHC system to ensure its reliability.

NARRATIVE

BACKGROUND

The Fort Calhoun Station (FCS) turbine is an 1800 rpm, tandem-compound, non-reheat unit with one high-pressure and two double-flow low-pressure turbines. Saturated steam is supplied to the turbine throttle from the steam generators through four stop valves (SV), numbered 1-4, and four control valves (CV), numbered 1-4. Steam flows through the high-pressure turbine and then through four moisture separators in parallel to two double-flow, low-pressure turbines, each of which exhausts to the condenser.

The turbine-generator control system (or electrohydraulic control (EHC) system) controls steam flow to the turbine. The control system consists of the following four parts:

a. Solid state controller and operator's panel;
b. Steam valve servo-actuator assemblies;
c. High pressure oil supply system;
d. Emergency trip or protection system.

The electronic controller performs basic analog computations on reference signals and turbine feedback signals and generates an output to the actuators. The operator's panel contains pushbuttons and switches which are used to change the reference input to the controller to vary the speed or load. Continuous monitoring of steam admission valve position, load limit setting and control signal is provided. The servo valves position the governing valves by directing the flow from the high pressure oil system to the actuators.

A loss of load reactor trip results from a turbine-generator trip at power levels greater than 15%.

The turbine-generator unit is controlled from the operator's panel in the control room. The panel indicates which devices are controlling the turbine-generator system. The turbine-generator control system is composed of solid state devices and servo-amplifiers which generate current, voltage and pulse-type signals.

EVENT DESCRIPTION

On March 13, 2008, the station was operating normally at 100 percent power. At approximately 1900 central daylight time (CDT) the output of the main generator was noted to be lowering. Control valve CV-3 was oscillating slightly. Power was reduced to a nominal 97 percent and the oscillation stopped. A troubleshooting plan was developed. On March 14 at about 1650 CDT power was gradually increased in accordance with the plan. Power was slowly increased without incident until March 15 at 0252 CDT when, at a nominal 100 percent power, control valves CV-1 and CV-3 moved slightly. Power was then reduced to a nominal 90 percent.

Following the shift and power reduction, the plant operated nominally at 90 percent power until 0406 CDT when a second movement of control valves CV-1 and CV-3 occurred. Power was lowered and stabilized at a nominal 85 percent at about 0440 CDT. The control valve oscillations stabilized with the power reduction to 85 percent. Further investigation and troubleshooting activities were initiated.

The plant operated at a nominal 85 percent power until 0833 CDT, when a reactor trip occurred. The operators entered Emergency Operating Procedure (EOP) 00 “Standard Post Trip Actions.” The main steam and feedwater systems operated normally. All control rods inserted fully. Following the trip, 2 of the 4 rod position indication systems indicated that 8 of the control rods were not fully inserted. The operators initiated emergency boration. The rods were later verified to have been fully inserted. The rod position indication problems were corrected prior to plant startup.

NARRATIVE

During the trip, the letdown system automatically isolated. The letdown system was manually restored at about 0903 CDT. A relief valve located between the containment isolation valves for the letdown system was noted to be leaking. (FCS LER 2008-002 will provide a detailed report of this situation.)

The emergency diesel generators (EDGs) automatically started as required on the reactor trip. EDG-1 was secured at 1007 CDT. EDG-2 was secured at 1020 CDT.

The operators transitioned from EOP-00 to EOP-02 “Loss of Off-Site Power / Loss of Forced Circulation,” due to the loss of power to a non-vital electrical bus during the trip. Power was lost to the non-vital bus due to a failure of the non safety related fast bus transfer function. At 0958 the operators entered the plant shutdown procedure, OP-3A. The plant was maintained in Mode 3 for the duration of the outage.

Discrepancies noted during the plant trip were entered into the station’s corrective action system for disposition. The corrective action document for the plant trip is CR 2008-1592.

At 1019 CDT, the NRC Headquarters Operations Office (HOO) was notified of the event per 10 CFR 50.72(b)(2)(iv)(B) and meeting the requirements of 10 CFR 50.72(b)(3)(iv)(A). This event is reportable per 10 CFR 50.73(a)(2)(iv)(A).

CONCLUSION

The cause of the reactor trip was the failure of a circuit board (AI50-B07) in the EHC control system resulting in the closure of turbine control valves CV-1 and CV-3. Closure of these valves results in a pre-emptive trip of the reactor due to the loss of load. A failure analysis of the circuit board has been completed. The analysis determined that the root cause of the circuit board failure was the improperly adjusted potentiometer R102, which resulted in a long-term overcurrent condition applied to the primary winding of transformer T201 on circuit board. This condition ultimately resulted in the failure of T201.

CORRECTIVE ACTIONS

The failed circuit board was replaced. The output of potentiometer R102 was verified to be correctly adjusted. Post maintenance testing was performed on the EHC system to ensure its reliability. Additional enhancements will be implemented by the station’s corrective action program.

SAFETY SIGNIFICANCE

Loss of Load is an analyzed plant transient and plant response was within the predicted response parameters. All control rods were inserted into the reactor core as required. Decay heat removal by the main steam and feedwater systems was available as required for the transient by plant procedures. While this event resulted in an actuation of the Reactor Protective System due to the Loss of Load from the turbine generator, it did not pose a threat to the health and safety of the public.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event does not result in a safety system functional failure in accordance with NEI-99-02.

PREVIOUS SIMILAR EVENTS

FCS has not had any previous similar reactor trips due to failures of the turbine control circuitry.